NON-PUBLIC?: N
ACCESSION #: 9505260304
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000220

TITLE: Reactor Scram Caused by Failure of Generator Protective
Relay
EVENT DATE: 04/19/95 LER #: 95-002-00 REPORT DATE:

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Mr. Kenneth Sweet, Manager
Technical Support NMP1 TELEPHONE: (315) 349-2462

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: FK COMPONENT: 21 MANUFACTURER: G080
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 19, 1995 at 1035 hours, Nine Mile Point Unit 1 (NMP1) received an automatic scram initiation signal resulting in a full reactor scram. The immediate cause of the scram was a trip of the Main Generator Lockout Relay (86G2 Relay) which tripped the turbine and resulted in a reactor scram. The 86G2 relay trip was initiated by failure of an oil-filled capacitor in a generator protective relay. At the time of the event, the plant was operating at 100 percent of rated thermal power. No testing or plant evolutions were in progress that resulted in or contributed to the scram.

Immediate operator actions included commencing scram recovery activities and initiating a controlled plant cooldown. Corrective actions included replacing the faulty relay. In addition, the generator protective relay design is being evaluated to determine appropriate measures to increase

reliability.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On April 19, 1995 at 1035 hours, Nine Mile Point Unit 1 (NMP1) received an automatic scram initiation signal resulting in a full reactor scram. The immediate cause of the scram was a trip of the Main Generator Lockout Relay (86G2 Relay) which tripped the turbine and resulted in a reactor scram. The 86G2 Relay trip was initiated by failure of an oil-filled capacitor in a generator protective relay. At the time of the event, the plant was operating at 100 percent of rated thermal power. No testing or plant evolutions were in progress that resulted in or contributed to the scram.

Following the scram, the non-safety, normal station 4160 volt powerboard 11 (PB 11) failed to automatically transfer to the reserve power source. The failure to transfer resulted in loss of some normal operating loads on PB 11. Manual operator action was immediately initiated in accordance with the applicable Special Operating Procedure (N1-SOP-1). PB 11 was manually restored within 45 seconds. Deviation/Event Report (DER) 1-95-1250 was issued to document the cause evaluation and corrective actions for the failure to automatically transfer.

Prior to the event, the Feedwater System was in a normal operating lineup consisting of Condensate Pumps 11, 12, and 13, Feedwater Booster Pumps 11 and 13, and Feedwater Pumps 12 and 13. Feedwater Booster Pump 12 was out of service for maintenance. Feedwater Pump 11 was in standby mode. Following the event, the Feedwater System transferred to the High Pressure Coolant Injection (HPCI) mode of operation. HPCI train 11 was initially unavailable, because Condensate Pump 11, Feedwater Booster Pump 11, and Feedwater Pump 11 are powered from PB 11. HPCI train 12, consisting of Condensate Pump 13, Feedwater Booster Pump 13, and Feedwater Pump 12 initiated, but Feedwater Pump 12 tripped on low suction pressure 8 seconds after the scram. The low suction pressure cleared 47 seconds after the scram, and Feedwater Pump 12 automatically restarted. Feedwater Pump 11 automatically started when power was restored and its start permissives were met at 49 seconds after the scram. Both pumps remained running in the HPCI mode until HPCI was reset at 52 seconds after the scram. Following the scram, reactor vessel water level recovered from a low of 37 inches indicated to approximately 105 inches indicated (the top of active fuel is at -84 inches indicated), with shaft-driven Feedwater Pump 13 providing the majority of this makeup during its coastdown.

The turbine trip from high power caused a pressure transient which resulted in five of the six Electromatic Relief Valves (ERVs) opening as expected, which caused torus water temperature to rise 1.5 degrees Fahrenheit. The longest duration of an ERV lift was four seconds.

The reactor scram occurred at 1035 hours on April 19, 1995, and the scram was reset at 1039 hours.

II. CAUSE OF EVENT

The immediate cause of this event was the failure of an oil-filled capacitor in the Loss of Excitation Relay (40 Relay) which resulted in a trip of the Main Generator Lockout Relay (86G2 Relay) causing a turbine trip and full reactor scram. Further testing of the capacitor is being conducted to determine the cause of its failure.

Immediately following the event, a walkdown of the generator protective relaying was performed. The Loss of Excitation Relay (40 Relay) was the only protective relay found to display a trip indication. A review of the generator protective relaying scheme determined that a trip signal from the 40 Relay will actuate the 86G2 Relay.

Also, the sequence of events log was reviewed and that information validates the conclusion that the 40 Relay was responsible for initiation of the 86G2 Relay.

Calibration procedure N1-RCPM-GEN070 was conducted on the 40 Relay in order to determine if the relay was operating properly. The relay failed the calibration test, and was removed from its case for further investigation of the problem. This investigation concluded that a defective capacitor in the relay was responsible for the failure. To confirm this conclusion, the capacitor was replaced and the relay passed the calibration procedure.

Visual inspection of the capacitor revealed no apparent cause of failure and it has therefore been sent to a testing laboratory for failure mode evaluation.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73(a)(2)(iv), "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

The full reactor scram was the design response of the RPS to a main turbine/generator stop valve closure with the plant at 100 percent power. The ERVs lifted, as designed, to prevent an overpressure condition. This reactor scram event was less severe and bounded by the Electrical Load (Generator Trip) Transient analysis in Chapter XV of the NMP1 Final Safety Analysis Report (FSAR).

The problem with the automatic transfer of PB 11 to reserve power did not adversely affect plant safety, although it added to the operator actions required to stabilize the balance of plant systems. The HPCI mode of the Feedwater System initiated as designed. Since the turbine shaft-driven Feedwater Pump 13 initially provides reactor vessel makeup, the initial unavailability of HPCI Train 11, along with the trip and subsequent restart of Feedwater

III. ANALYSIS OF EVENT (Cont'd.)

Pump 12, did not affect the severity of this event. The loss of PB 11 resulted in only one Condensate and one Feedwater Booster Pump running, and, with shaft-driven Feedwater Pump 13 consuming most of the flow, contributed to the low suction pressure trip of Feedwater Pump 12. Both Feedwater Pumps restarted when their start permissives were met, and they continued to operate in the HPCI mode until they were no longer needed.

This event had no adverse consequences. It did not adversely affect any other safety system nor the operators' ability to maintain safe reactor plant conditions. This event in no way adversely affected the safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

The immediate corrective actions were for operators to perform the scram recovery actions, place the plant in a stable condition, and determine the cause of the scram.

Additional corrective actions include:

1. A Deviation/Event Report (DER 1-95-1236) was issued to track the evaluation of the event, the LER, and any corrective actions.

2. The faulty 40 Relay was replaced. The original relay is no longer

manufactured; a new, but functionally identical model was installed.

3. The failed capacitor has been sent to a testing laboratory for further failure mode evaluation. Depending upon the results of the failure mode evaluation, additional appropriate actions will be taken to minimize the potential for similar failures of other relays that can, by design, singularly cause a protective trip. As a contingency measure, the Relay and Control and Engineering Departments conducted a review of distance relays and identified those which could cause a generator trip. Niagara Mohawk is currently identifying and obtaining replacement and spare parts for selected relays, should rebuilding or replacing relays be determined to be an appropriate preventive action.

In addition, as part of the scram reduction program, the generator protective relaying design is being evaluated to determine appropriate measures to increase reliability. This evaluation had been included in the Nuclear Business Plan and is schedule to be completed by mid-year 1995. The evaluation will focus on components where a single failure can result in a reactor scram. This evaluation is expected to result in changes to the design and/or maintenance of protective relays in order to increase plant reliability.

V. ADDITIONAL INFORMATION

A. Failed component:

GE Relay Model Number 12CEH11A3A

B. Previous similar events:

LER 94-002 addressed a reactor scram caused by a failure of a line protective relay. For the previous event, the failed relay
(distance
irectional) and type of capacitor (electrolytic) were different than the current event. Corrective actions for the previous event included replacing the faulty relay and another similar line protective relay with available spares. These two relays were subsequently replaced with newer models during the last refueling outage (RFO-13 completed in April 1995).

C. Identification of components referred to in this LER:

COMPONENT IEEE 803 FUNCTION IEEE 805 SYSTEM ID

Reactor Protection System N/A JC

Turbine Generator TRB TA

Main Feedwater System N/A SJ

High Pressure Coolant Injection
System N/A BJ

Reactor Pressure Vessel N/A SB

Protective Relay 21 FK

Power Board JX EA

Pumps P SJ

Relief Valve RV SB

ATTACHMENT TO 9505260304 PAGE 1 OF 1

NIAGARA
MOHAWK

NINE MILE POINT NUCLEAR STATION/P.O. BOX 63, LYCOMING, NEW YORK 13093/TELEPHONE (315) 343-2110

May 19, 1995
NMP1L 0949

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Docket No. 50-220
LER 95-02

Gentlemen:

In accordance with 10CFR50.73 (a)(2)(iv), we are submitting LER 95-02, "Reactor Scram Caused by Failure of Generator Protective Relay."

Very truly yours,

R. B. Abbott

Plant Manager - NMP1

RBA/AFZ/kab
Attachment

xc: Mr. Thomas T. Martin, Regional Administrator, Region I
Mr. Barry S. Norris, Senior Resident Inspector

*** END OF DOCUMENT ***